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SOME PHYSICS CALCULATIONS ON THE PERFORMANCE OF FAST POWER REACTORS

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In the early stages of fast power reactor development the study of initial data variation effects acquires great importance as well as the discovery of means to promote their performances and analysis of major reactor effects. The reactor comparison goes multiple ways. In the present report reactors are compared from the standpoint of the opportunities available for attaining the least possible time required for doubling power level in the fast reactors system. To strike this goal for power reactors of 0.5 to 1.0 million kW electric capacity range the following conditions seem indispensable.

a) A sufficiently high breeding ratio is maintained. To avoid considerable reductions of the breeding ratio as reactor capacity goes up the extensive use can be made of various technological solutions - heat generation flattening, reshaping of the reactor core, etc.

b) The fast reactor fuel must allow high burn-ups. As it seems, ceramics are most farreaching herein, besides, it have good technological properties.

c) The specific power per a unit of fuel loading must be reasonably high. The latter calls for an intensive heat removal. Sodium used as the coolant provides for the intensive heat removal under ordinary working conditions together with the reliable accidental cooling. Once sodium occurs in combination with ceramic fuel it becomes possible to operate at high temperature rates with attractive atomic station's efficiency factor.

This report deals with the outcomes of a study programme regarding fast power reactors with ceramic fuels and sodium coolant.

The Multiplication Equations for the System of Breeders. To regard the different variants of nuclear power developments one needs an economic comparison. The latter begins with defining the natural production volumes to provide for the above nuclear energy developments - those of uranium output, enriching industry, chemical processing etc. The following equations are helpful for these ends, and so they are for obtaining the formula for the nuclear energy industry's specific rate of growth on the basis of the fission materials developed in the system of reactors under consideration. The specific rate value or doubling time value determines to a great extent the demand for uranium-235 and other characteristics at the given power industry development rate

It is necessary for the reason of generalization to assume that some three reactor types are employed to promote advance in the nuclear power production: those on uranium-235, plutonium-239 and uranium-233. Let $P_5(t)$, $P_3(t)$, $P_9(t)$ denote the outside daily consumption of uranium-235, uranium-233, plutonium-239 at some t moment. The possibility of channelling fission materials for other purposes makes $P_{(t)}$ presumably acquire negative values. $i_5(t)$, $i_3(t)$, $i_9(t)$ constitute a daily fuel U^{235}, U^{233}, Pu^{239} load in reactors at the t moment of time. For the sake of convenience it is suggested to consider the quantity M_i of the given isotop already in the fuel elements at the time of loading into the reactor, and yielding the predetermined power level, for instance of 1000MW, to be a unit of fuel. Meanwhile, the units of fuel for U^{235}, Pu^{239}, U^{233} reactors differ to make M_5, M_9, M_3 respectively. Every reactor core fuel assembly is expected to be discharged on attaining the permissible value of burnup and remains within the reactor core the average time of T_a-the reactor lifetime. Besides, it subsequently undergoes chemical processing for the time of Tna, here a loss of fuel makes (1-E). Thus, the daily recurrence into uranium reactor cores at the t moment is $(t-Te_5-Tne_5)\ [1-\Delta_5]\ E_5$ of uranium-235. Here, Δ - is the fuel burnup due to fission and capture in the reactor core as regards the original fuel load.

The breeding of the secondary nuclear fuel "m" in the reactor, utilizing the i-th isotope, can be described in terms of the reactor core breeding ratios alongside with those of axial and radial blankets of the reactor: $BR_{a\ell,m}$, $BR_{\partial r\ell,m}$, $BR_{\partial\ell,m}$.

Pu-239 produced in the U^{235} reactor core and the axial and radial blankets at the t moment of time enters plutonium reactors in the quantities:

$$i_5(t-T_{a5}-T_{na5})\Delta_5 KB_{a5,9}\varepsilon_9 + i_5(t-T_{a5}-T_{n\partial5})\Delta_5 KB_{\partial r5,9}\varepsilon_9 \qquad (1)$$

The following model is accepted to describe the inlet of plutonium from the radial blanket.

It takes some $T_{\partial5}$ time since the reactor gets working before the uniform discharge of plutonium in the quantity of $\frac{\Delta_5}{T_{a5}} KB_{\partial65,9}$ begins and the addition of $T_{n\partial5}$ time before it enters the reactor. The number of the reactors concerned by the time t constitutes the integral of all the fuel to have been used for the period since the first reactor was commissioned:

$$n_5(t) = \int_0^t [i_5(t') - i_5(t'-T_{a5})]\,dt' \qquad (2)$$

The supply of plutonium (kg) within the unit of time from radial blankets of U^{235} reactors makes:

$$M_5 \frac{\Delta_5}{T_{a5}} KB_{\partial65,9} \int_0^{t-T_{\partial65}-T_{n\partial5}} [i_5(t') - i_5(t'-T_{a5})]\,dt' \qquad (3)$$

Expressions for other types of reactors are recorded similarly. It must be borne in mind, that very same quantities of uranium-235, uranium-233, plutonium-239 are not equally effective in yielding similar powers that is reflected in different values for M_i.

Thus, the balance equations of U^{235}, Pu^{239} and U^{233} take up the form:

uranium-235

$$i_5(t) = i_5(t-T_{a5}-T_{na5})(1-\Delta_5)E_5 + P_5(t) \qquad (4)$$

plutonium-239

$$\frac{1}{E_9} M_9 i_9(t) = M_5\Big\{ i_5(t-T_{a5}-T_{na5})\Delta_5 KB_{a5,9} + i_5(t-T_{a5}-T_{n\partial5})\Delta_5 KB_{\partial T5,9} +$$
$$+ \frac{\Delta_5}{T_{a5}} KB_{\partial65,9} \int_0^{t-T_{\partial65}-T_{n\partial5}} [i_5(t') - i_5(t'-T_{a5})]\,dt'\Big\} + M_9 i_9(t-T_{a9}-T_{na9})[1-\Delta_9(1-KB_{a9,9})] +$$
$$+ M_9 i_9(t-T_{a9}-T_{na9})\Delta_9 KB_{\partial T9,9} + M_9 \frac{\Delta_9}{T_{a9}} KB_{\partial69,9} \int_0^{t-T_{\partial69}-T_{n\partial9}} [i_9(t') - i_9(t'-T_{a9})]\,dt' +$$
$$+ M_3\Big\{ i_3(t-T_{a3}-T_{na3})\Delta_3 KB_{a3,9} + i_3(t-T_{a3}-T_{n\partial3})\Delta_3 KB_{\partial T3,9} +$$
$$+ \frac{\Delta_3}{T_{a3}} KB_{\partial63,9} \int_0^{t-T_{\partial63}-T_{n\partial3}} [i_3(t'-T_{a3}) - i_3(t'-T_{a3})]\,dt'\Big\} + P_9(t)\frac{M_9}{E_9} \qquad (5)$$

uranium-233

$$\frac{1}{E_3} M_3 i_3(t) = M_5\Big\{ i_5(t-T_{a5}-T_{na5})\Delta_5 KB_{a5,3} + i_5(t-T_{a5}-T_{n\partial5})\Delta_5 KB_{\partial T5,3} +$$
$$+ \frac{\Delta_5}{T_{a5}} KB_{\partial65,3} \int_0^{t-T_{\partial65}-T_{n\partial5}} [i_5(t') - i_5(t'-T_{a5})]\,dt'\Big\} + M_9\Big\{ i_9(t-T_{a9}-T_{na9})\Delta_9 KB_{a9,3} +$$
$$+ i_9(t-T_{a9}-T_{n\partial9})\Delta_9 KB_{\partial T9,3} + \frac{\Delta_9}{T_{a9}} KB_{\partial69} \int_0^{t-T_{\partial69}-T_{n\partial9}} [i_9(t') - i_9(t'-T_{a9})]\,dt'\Big\} +$$
$$+ M_3\Big\{ i_3(t-T_{a3}-T_{na3})[1-\Delta_3(1-KB_{a3,3})] + i_3(t-T_{a3}-T_{n\partial3})KB_{\partial T3,3}\Delta_3 +$$
$$+ \frac{\Delta_3}{T_{a3}} KB_{\partial63,3} \int_0^{t-T_{\partial63}-T_{n\partial3}} [i_3(t') - i_3(t'-T_{a3})]\,dt'\Big\} + P_3(t)\frac{M_3}{E_3} \qquad (6)$$

The system of equations (4-6) makes it possible to analyse the efficiency of various kinds of reactors, and also that of whole systems of reactors of various kinds. In the general case, the solution of system of equation is accomplished numerically from minor times to major

ones. In the case, however, when the rate of growth of nuclear electric generating capacity is given the equations serve to determine major natural production volumes for the development process under consideration.

The very same equations are available for determining the specific rate of growth. For instance, the assumption of plutonium breeders only necessitates the use of equation (5). The solution of equation (5) is to be sought for in the form $i_9(t)=C_0 e^{\omega t}$ where ω - is the specific rate of growth. The substitution of this solution in equation (5) offers characteristic equation for the purpose of defining ω :

$$\frac{1}{\varepsilon_9} = e^{-\omega(T_{a9}+T_{na9})}\left[1-\Delta_9(1-KB_{a9,9})\right] + \frac{\Delta_9(1-e^{-\omega T_{a9}})}{\omega T_{a9}} KB_{35,9,9}\, e^{-\omega(T_{359}+T_{n39})} + \Delta_9 KB_{3T9,9}\, e^{-\omega(T_{a9}+T_{n39})} \tag{7}$$

A formula with a satisfactory degree of approximation to provide for the doubling time ($T_2=\ln^2/\omega$) is obtained by expanding the exponent (4) in series with the accuracy to the first powers ω :

$$T_2 = \frac{\ln 2\left[\Delta_9(KB_{9,9}-1) - \frac{1-\varepsilon_9}{\varepsilon_3}\right]}{\Delta_9 KB_{359,9}(T_{359}+T_{n39}) + \Delta_9 KB_{3T9,9}(T_{a9}+T_{n39}) + \left[1-\Delta_9(1-KB_{a9,9})\right](T_{a9}+T_{na9})} \tag{8}$$

Equations (7) and (8) are applicable as well for the definition of the doubling time in the cycle U^{233} - Th^{232}. The combined solution of equations (5) and (6) permits the definition of the doubling time in the reactor system on Pu^{239} and U^{233} (it is suggested that uranium-233 is bred in blankets only). The solution for the equation system (5) to (6) is obtainable in the form:

$$i_9(t) = a e^{\omega t} \qquad i_3(t) = b e^{\omega t}$$

The introduction of the uniform ω means that the mixed system developes at equal rate while the relation between the numbers of reactors on U^{235} and Pu is constant throughout the process of development. The corresponding characteristic equation for ω has the form:

$$\frac{1}{\varepsilon} - (1-\Delta_3)e^{-\omega(T_{a3}+T_{na3})} - \frac{1-e^{-\omega T_{a3}}}{\omega T_{a3}}\Delta_3 KB_{353,3}\, e^{-\omega(T_{353}+T_{n33})} - \Delta_3 KB_{3T3,3}\, e^{-\omega(T_{a3}+T_{na3})} = \frac{\Delta_3 KB_{a3,9}\, e^{-\omega(T_{a3}+T_{na3})}}{\frac{1}{\varepsilon}-\left[1-\Delta_9(1-KB_{a9,9})\right]e^{-\omega(T_{a9}+T_{na9})}} \cdot \left[\Delta_9 KB_{3T9,3} \times e^{-\omega(T_{a9}+T_{na9})} + \frac{1-e^{-\omega T_{a9}}}{\omega T_{a9}}\Delta_9 KB_{359,3}\, e^{-\omega(T_{359}+T_{n39})}\right] \qquad (\varepsilon_9=\varepsilon_3=\varepsilon) \tag{9}$$

The following expressions obtained for the coefficients a and b

$$a = \frac{\omega}{1-e^{-\omega T_{a9}}} \qquad b = \frac{c\omega}{1-e^{-\omega T_{a3}}}$$

where $C=\frac{n_3(t)}{n_9(t)}$ is established on the basis of the accepted ratio between the numbers of Pu and uranium-233 reactors.

The above mentioned numbers of the reactors are determined by the expressions of the type (2). Finally

$$C = \frac{M_9}{M_3}\cdot\frac{1-e^{-\omega T_{a3}}}{1-e^{-\omega T_{a9}}}\cdot\frac{\frac{1}{\varepsilon}-\left[1-\Delta_9(1-KB_{a9,9})\right]e^{-\omega(T_{a9}+T_{na9})}}{\Delta_3 KB_{a3,9}\, e^{-\omega(T_{a3}+T_{na3})}}$$

Used Cross-Sections and Calculation Methods.

In large fast reactors the neutron spectrum in the reactor core and blankets is softened because of the neutrons slow-down by U^{238} oxygen or carbon, construction materials and coolants.

In many processes neutrons below the 10 kew energy level cause essential effects. The fraction of fission below 10 kev in ceramic fuel reactor constitutes 10-30% depending on the size and composition of the reactor. This makes it necessary to approach most attentively group constants in the region of energies below 10 kew where the significance of resonance self-shielding effects with regard to the composition of the medium is obvious.

Besides, the contribution of elastic slow-down must be carefully considred, for group cross-sections of elastic slow-down are dependent to no small extent on the group's internal spectrum. Therefore it needs to bring the spectrum and slow-down cross-sections to precision one after another. The 26-group nuclear data set worked out under I.I.Bondarenko 1 takes into consideration all these peculiarities in the most complete form. Before that cross-section set was accomplished, another 18-group system in practical use for fast reactors calculations. The latter was produced with a special eye to the spectrum of oxide and carbide reactors with the reactor core volume approximately 2000 litres and the composition as follows: 30-40% of fuel, 20-25% of stell, the rest goind to sodium.

In that set for the region of energies above 10 kev the cross-sections from the reference 2 were used as the foundation. In the range of energies below 10 kev groups were made, while the region above 1,4 Mev fell into four groups. The former being adapted to the effects of resonance self-shielding. The cross-sections were produced by I.I.Bondarenko, L.P.Abagyan, N.O.Bazazyants and S.B.Shikhov. The calculations in line with this system as well as the 26-group system for the reactors of the above composition give approximately the same results.

The major physical calculations were accomplished for the one-dimensional geometry in terms of the diffusion approximation. The reactor's critical parameters, the energy-space distribution of neutrons and their importance, interrelations among various process-numbers and breeding ratio were computed on the basis of the programmes for digital computation worked out by G.I.Martshuck, V.P.Kotshergin, A.I.Nevinitsa, T.I.Zuravleva and others.

Quite a number of similar programmes were developed and perfected by A.I.Shmelev, I.S.Slesarev, A.N.Kuzmin, V.V.Chromov, S.B.Shikhov. The calculation programme on reactivity fluctuations depending on the introduction of various perturbations into the reactor core was produced by M.N.Zizin.

In some cases the P-3 approximation (the programme by G.I.Martshuck, E.I.Lyashenko, L.I. Kuznetsova) was used to realise the accuracy of diffusion calculations. The comparative computations under the P-3 and diffusion approximations for the reactor cores radius of 60-110 sm and the blanket 40-60 sm thick, resulted in the coincidence in K_{eff} with the accuracy of 0.5%. The space distributions of a neutron field under these approximations are also in good agreement along the whole reactor radius.

THE PHYSICAL CHARACTERISTICS AND DOUBLING TIME FOR REACTORS OF VARIOUS POWERS AND POWER DENSITIES

The fact that the experimental reactor BR-5 whose specific characteristics are very close to those of a large power reactor operated very effectively paved the way for designing the BN-350 type reactor, discribed in the report (4).

Below the results are quoted of the reactor survey for reactors of different powers capacities and densities power with the initial parameters close to those of the BN-350 type. The initial data to enter the calculations were as follows:

Sodium inlet temperature	300°C	
Sodium inlet-outlet temperature difference	230°C	
Maximum sodium velosity	10 m/sec	
Fuel clad thickness (stainless steel)	0.4 mm	
Volume fraction of the assembly walls and interassembly Na		0.14
Reactor core diameter to height ratio (D/H)		1.4

The assembly walls interassembly Na and volume fraction Ca 0.4% reactor core diameter to height ratio (D/H) = 1.4.

The value D/H = 1.4 stands out as the optimum for BN-350 reactors and is acceptedly equal for all the reactors dealt with in this section. It will be shown later on that under other pre-conditions the shape of the reactor core has the optimum for somewhat different values of D/H.

The dioxide (PuO_2-UO_2, = 8 g/cm^3) and monocarbide (PuC-UC, = 10 g/cm^3) were considered as the fuel materials for this group of reactors. The oxide or the carbide of depleted uranium played the blanket substance, the latter being 60 cm thick.

Tables I and II carry the basic data for the reactors in question in relation to the pre-given power and power density. Reactor comparison along the fuel principles shows the following results. A higher density and a greater heat conductivity of the carbide fuel provides for more U^{238} to be placed in the reactor core. This brings about an increase of the contribution of U^{238} fission. The spectrum in the carbide reactor cores is somewhat more hard than that of oxide reactors. As a result, the breeding ratio of carbide reactors is 1.45-1.8, while oxide reactors have it 1.35 - 1.6. However, the critical mass in carbide reactors is greater, thanks to the more intensive neutron absorption in uranium-238.

Within the considered alterations in the reactor core composition for the reactor core radii below 120 cm the reactor critical masses with 5-7% precision satisfy the expression $G \quad V_a$ inherent in the similarity of reactors 6 .

The power being constant, the reactor composition depends to a great extent on the specific power and so do the critical mass and internal breeding ratio. The full breeding ratio (BR) is not open to sharp changes.

The neutron spectrum under permanent power conditions alters only very slightly with the variations of specific power, though there is a slight "broadening" of the spectrum as long as the reactor core size diminishes. The broadening has a bearing on the decrease of the non-elastic scattering of fast neutrons, and the relaxed moderated neutron absorption with diminishing quantities of uranium-238. In general, it can be seen that the macroscopic characteristics (critical mass, BR, U^{238} fission fraction) to be dependent not on the spectral alterations alone but furthermore on the varying fuel contents in the reactor core.

The increase of reactor power has a notedly bad effect on its physical performances. The volume contents of fuel drops, the U^{238} fission reduces, the neutron slow-down by sodium softens the spectrum considerably. As a result, the BR goes down.

In doubling time calculations the following additional assumptions were made.

The out-of-reactor fuel residence time T_n.....0.5 year

Maximum burn-up -in the oxide-10%, in the carbide - 8%

Fuel losses in the course of processing 1.5%

Average build-up of plutonium in the fuel discharged from the radial blanket 1%

To evaluate the effect of the fuel cycle parameters the doubling time was calculated for other cycle parameters.

The off doubling time may have the minimum values for some intermediate values of Q as long as the reactor power is constant and its specific power Q alters. The above minimum is linked up with the various ratio of the reactor's lifetime and the external cycle duration for different Q-s and with the less critical mass for increasing Q.

For the reactor parameters under consideration optimum doubling time increases slightly with the increase of reactor power. It follows from the BR drop with power increasing; this drop is not compensated for by the increase of specific power per a unit of plutonium loading.

It was studied, how separate parameters of a fuel cycle influence the doubling time. The effect of the value of burn up on the doubling time is twofold. On the one hand, burn-up increasing the processed fuel fraction drops as does the frequency of discharges, that causes the reduction of the doubling time. On the other hand, the concentration of fission produots increases, breeding ratio goes down and the doubling time increases. These factors open possi-

bilities of the exixting optimumon the value of burn-up.

For the oxide fueled reactor (900MW, 500 kW/l) with fuel burn up 5% doubling time is 8,5 year, with burn-up 10% - 6,5 year and 6,1 year, when burn-up reaches 15%. The doubling time has it's minimum value with fuel burn-up 16%, further increase in burn-up causes increase of the doubling time.

The results of the comparison between the reactors under view as to the doubling time show that the accepted initial parameters ensure the doubling time of 6 to 8 years for power reactors. Possible ways to increase breeding ratio and specific power as well as to bring down doubling time, are considered in bring the next section.

POSSIBLE IMPROVEMENTS IN THE PERFORMANCE OF LARGE FAST POWER REACTOIS.

It may be well suggested, that certain parameters are probable to be improved upon in advanced fast reactors. As it seems, the core temperature drop and velocity of sodium could be enhanced. This increases fuel contents in the reactor core and diminishes quantity of sodium. As a result, the fuel residence time in reactor becomes longer, parallel to an increase of BR and slump shortening of the doubling time.Perhaps,the outlet temperature of sodium can be raised to make 600-630°C and higher efficiency can be achieved.It is as probable,that ceramic fuel yields a burn-up of approximately 15%.

Besides, the reactor core may need specific solutions with a view of improving physical parameters of large reactors. A possible solution offers the flattening of the heat generation in radial direction playing to bring down the (not spot factor). The reduction of the radial heat non-uniformity makes it possible to either bring down the size of the reactor, while preserving the marimum powbr density and composition, or enhance the fuel contents and make the fuel element diameter increased, the reactor size and average power density being just what they were. There might be, of course, certain in-between solutions. Properly speaking, the heat flattening may go different ways:

1) a higher concentration of fuel from the centre sideward;
2) the establishment of internal blankets.

The first possibility bears greatest fruit in case of the radial zones with fuels of various fissionable material concentration in the fuel (various enrichments). It is far less effective to change the fuel element's lattice, or its diameter with permanent enrichment.

It appears that as few as two zones of different enrichment can change the radial non-uniformity coefficient in large reactors from 1,7-1,8 to 1,2-1,3. The volumes of the zones with different enrichments are approximately equal ($V_1 \cong 0{,}52\ V_a$) and this relation is weakly dependent on the reactor size and composition.

Heat generation flattening by enrichment affects physical characteristics as soon as there is a different fuel content, the spectrum hardening in the reactor core, neutron leakage via a radial blanket, particularly, in its high energy region. The relative importance of the above effects varies in different reactors, being dependent on the kind of fuel, the size and composition of the reactor. Yet, in all the above mentioned cases the total breeding ratio increases with flattening. Comparisons of reactors with and without flattening are shown in Table III.

It is evident that flattening may allow of a considerable reduction of the volume and critical mass of a reactor with a pre-set power level.Such a use of flatteningif coupledwith a deep burn-up (15%) and semi-annual outer cycle and also a smaller-size reactor core brings about a substantial cut of the doubling time.

If one is far less enthusiastic about the fuel cycle (β=5%, T_c= year) it should be preferrable to use flattening to enlarge fuel volume fraction in the reactor core. Though it results in an increase of the critical mass, yet the fuel contents in the cycle does not increase because of a longer lifetime, the breeding ratio goes up and, finally, the doubling time shortens as compared with the case of a reactor without flattening.

Possibilities to flatten power distribution by setting up an internal cylindrical ring breeding zone were studied, and the selection of the internal breeding zone size and its location on the condition of the least doubling time took place. In all of these cases the reactor core assemblies were of the same composition and enrichment.

Table III gives results of calculations for such reactors. The introduction of 5-10% (toward the reactor core volume) assemblies with U^{238} appears to be able to diminish the radial non-uniformity of power distribution just like the case of zones with different enrichment. This effect is capable of cutting down the doubling time by 15-20%. Very similar results can be arrived at if assemblies with depleted uranium are placed in the center of the reactor core in addition to fuel assemblies.

A substantial improvement of physical characteristics is attainable by giving the fast power reactor core the form of a flattened out cylinder. A rise in D/H in a reactor of a given power and constant power density allows to enlarge the volume fraction of fuel, prolong the core's lifetime and increase breeding ratio. Because of a greater D/H and a simultaneous increase of the critical mass there must be the optimum D/H for every reactor[x].

The optimum D/H value goes up with the power density increase. For example, for a carbide reactor with thermal power of 2500 MW, and flattened heat generation the optimum D/H defined in accordance with the doubling time made 1,5 at $\bar{q} = 800 \frac{kw}{litre}$, and 2,5 ÷ 3 at $\bar{q}$ 1200 $\frac{kw}{litre}$.

The position of the optimum D/H is in close connection with the fuel volume fraction and core lifetime, on the one hand, and on the other hand, with the critical mass and breeding ratio. In high power density reactors the fuel volume fractions are low and so is the core lifetime which both favour flattening out to far greater D/H than in reactors of lower (power density) where the breeding ratio and core lifetime are high enough even at moderate D/H. For similar reasons the optimal D/H goes up as long as the reactor power enhances. The least doubling time for each power level might be achieved with some moderate power densities and corresponding optimal D/H.

For the reactor in question the minimal doubling time occurs at $\bar{q} \cong 1000$ kw/litre and D/H 2-2,5, and makes approximately 3 years (the processing time 0,5 year, burn-up 15%).

The implementation of all the suggestions mentioned herein (high power density, power flattening, and increased D/H) opens the road towards substantially improved performance of large fast power reactors (electric power $\sim$ 1000 MW). For this and still more powerful types of reactors on ceramic fuel with high burn-up one can hope for the doubling time below 5 years: for oxide fuel $\sim$ 4 years, for carbide fuel $\sim$ 3÷3,5 years.

Temperature and Power Effects

Reactivity changes during reactor transient are characterized by power and temperature coefficients (MWt^{-1} and $°C^{-1}$, respectively for uniform and non-uniform heating) and are determined by the following processes:

a) Changes in the geometrical sizes of reactor. The temperature elongation of fuel elements and expansion of grids in which the assemblies are fixed lead invariably to negative reactivity effects. For the considered reactor type it is possible to work out the following approximate relations:

$$\Delta K/K = -(0,75 + 0,9)\ \Delta V/V$$
$$\Delta K/K = -(0,5 + 0,6)\ \Delta R/R$$
$$\Delta K/K = -(0,25 + 0,3)\ \Delta H/H$$

$\Delta V/V$, $\Delta R/R$, $\Delta H/H$ here represent a relative changes of volume, radius and height of a reactor.

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x) The problems of the kind were tackled by A.M.Kuzmin, I.S.Slesarev, V.V.Khromov, S.B.Shikhov, A.N.Shmelyov (in the press).

b) Changes in the sodium density. The accompaning reactivity changes are primarily related to changes in the spectrum and redistribution of captures, fissions and leakage processes. In moderate-size reactors reductions of sodium density yield negative effects. For the reactors, where the leakage fraction is small in a neutron balance a positive effect of the sodium density decreasing might be the case. The critical volume of reactors to make the effect sign change is ascertained through the type of fuel and the reactor design peculiarities . The sodium coefficient becomes more positive for an established volume, if absorbing materials are introduced into the reactor core (boron control rods, for instance). In the same way acts power flattening in the reactor core.

Fig.1 gives an example of the reactivity changes due to introduction of 1 cm^3 of sodium into various types of reactors.

c) Doppler-effect. Doppler-effect may constitute a large part of the overall power effect in large size reactors with ceramic fuel. Uranium-235 and plutonium-239 being heated up give positive effect, that of uranium-238 being negative.

When U^{238} is used in a reactor as a diluent, the contribution of the uranium to Doppler-effect is affected to a great extent by the reactor core size. As soon as the volume enlarges so does the concentration of uranium-238 (with a constant fuel fraction), and the resonance neutron fraction as well (10-0,005 Kev). In the reactors under consideration (1500-2000 litres, the negative contribution of uranium-238 into Doppler-effect is a determining factor. Studies(1) and [5] carry the cross-section data to calculate Doppler-effect. Fig. 2 shows how Doppler-effect depends on the temperature in a reactor of $\sim$ 2000 litre volume (the fuel PuO_2-UO_2). Fig.3 gives the effect's distribution among energy-groups for the same reactor. The calculation was made on the basis of the perturbation theory.

d) Fuel elements and assembly bendings. The effects of these are primarily determined by the reactor design, their contribution being comparatively small in the case of large reactors.

Table IV shows the component constituents of power and temperature effects for one of considered reactors.

Investigation of Possibilities to Use Thorium in Fast Power Reactors

A number of data available at present on the cross-sections of U^{233} and Th^{232} [I] show that breeding of U^{233} is handy on either thermal or fast neutrons. In the intermediate energy range (1 ev-Ikev) ν_{eff} for U^{233} is below 2, which excludes any chance of breeding in intermediate reactors. In the U^{233}-Th^{232}-U^{233} fuel cycle the breeding ratio in thermal reactors amounts to I.I, to become I.2 - 1.3 in fast reactors depending on the fuel. Doubling times make 15-20 years. Such figures look far inferior than in the Pu^{239}-U^{238}-Pu^{239} cycle of fast reactors. Still, the combination of this two cycles in fast reactors alongside with a mixed cycle could essentialy improve the prospects of using thorium in the atomic power industry.

The nuclear fuel of the mixed cycle is U^{233} and Pu. The fertile material for reactor cores is U^{238}, while thorium is used for blankets. Under this system the neutron balance subsantially improves as compared to the cycle of U^{233}-Th through a more effective fission of U^{238} and also at the expence of high ν_{eff} for Pu^{239} in the reactor core spectrum.

The implementation of mixed cycles in actuality may be different. U^{233} and plutonium may locate in a single reactor as well as in different ones. In a single reactor it can be either a homogeneous fuel mixture or else a location in different fuel elements or different assemblies. The comparison between mixed and single cycles for different reactors manifests the following mixed cycle physics peculiarities.

The critical mass of mixed fuel reactors, if summirized after uranium-233 and plutonium-239 is 10% less than that in simple cycles. This can be attributed to a more effective use of U^{233} in the mixed reactor spectrum. The value $\eta=(\nu-1-\alpha)\sigma_f$ serves a true indicator of this, it being $\eta_3/\eta_9 \simeq 1.4$ in an ordinary reactor core spectrum.

The breeding ratio of a mixed cycle is determined (as far as a mixed fuel reactor) as a

ratio of the summarized production rates of U^{233} (in blankets) and Pu^{239} (in a reactor core), to the burning rates of these isotopes in the reactor core. The mixed systems' breeding ratio in the case of an oxide fuel is 0.1-0.3 higher than in the cycle U^{233}-Th. The location of thorium in the blanket makes changes of BR because of the neutron capture in protactinium becomes negligible. The doubling time in a mixed cycle appears to be far below that in the cycle of U^{233}-Th (Table V).

Meanwhile, the developing mixed system allowes an intensive use of thorium in place of uranium-238. Clearly, a developing system necessitates the demand for nuclear fertile materials as long as it calls for new blankets. In the mixed cycle thorium-loading in blankets is 10 (ten) times more than that of uranium-238 into reactor cores, which is, in fact, a determining factor of the fertile material consumption rate in a new reactor construction. Definitely, it depends to no small extent on the kind of fuel and other specific features of a reactor. As for the reactor alone, the fertile material demand is determined by the rate of burning and processing losses only. Besides the advantages mantioned above the mixed cycle opens up a possibility of manufacturing U^{233} of a less activity on the basis of U^{232}.

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Table I

Characteristics of PuO_2-UO_2 reactors

Reactor core power, MW	450				900				I500			2250		
Power density, kw/litre	300	500	750	900	300	400	500	600	300	400	500	300	400	500
Volume fractions: fuel	0.548	0.474	0.396	0.368	0.470	0.409	0.362	0.3I2	0.419	0.354	0.302	0.330	0.298	0.I9I
Volume fractions: sodium	0.294	0.292	0.352	0.377	0.3I2	0.362	0.404	0.450	0.366	0.424	0.472	0.464	0.485	0.605
Volume fractions: steel	0.2I8	0.234	0.252	0.255	0.2I8	0.229	0.234	0.238	0.2I5	0.222	0.226	0.206	0.2I7	0.204
Critical mass of plutonium, kg	7I0	460	340	290	II70	880	7I0	600	I7I0	I260	I000	2I80	I650	I060
BR	I,60	I.60	I.55	I.56	I.52	I.48	I.46	I.44	I.45	I.40	I.38	I.36	I.34	I.24
BRa	0.9I	0.68	0.46	0.40	0.94	0.79	0.68	0.56	0.97	0.8I	0.68	0.90	0.80	0.54
Fission fraction U-238 in reactor core	0.I53	0.I80	0.I00	0.093	0.I45	0.I28	0.II3	0.097	0.I39	0.I20	0.I03	0.I2I	0.I09	0.074
Reactor core average: $\alpha = \frac{\sigma_{cs}}{\sigma_{fs}}$	0.2I7	0.2I2	0.208	0.208	0.234	0.236	0.237	0.239	0.248	0.253	0.257	0.270	0.273	0.297
Reactor core average: σ_{fg}	I.92	I.92	I.92	I.9I	I.96	I.97	I.98	I.99	2.0	2.02	2.03	2.06	2.07	2.I6
Reactor core average: σ_{fs}	0.048	0.05I	0.055	0.05I	0.044	0.045	0.046	0.047	0.042	0.043	0.043	0.040	0.040	0.04I
Reactor core average: σ_{cs}	0.287	0.28I	0.275	0.270	0.303	0.304	0.304	0.304	0.3I6	0.3I9	0.32I	0.335	0.337	0.356
Blanket average: σ_{cs}	0.384	0.38I	0.377	0.375	0.390	0.389	0.388	0.387	0.394	0.394	0.394	0.400	0.400	0.403
Blanket average: σ_{fs}	0.0I8	0.0I9	0.020	0.020	0.0I8	0.0I8	0.0I8	0.0I8	0.0I7	0.0I7	0.0I7	0.0I7	0.0I7	0.0I7
Doubling time, years $T_{pr.}$ 0.5 y β = 10%	8.6	6.5	5.6	5.7	8.I	7.0	6.5	6.6	8.5	7.6	7.2	9.0	8.3	9.9

Table II

Characteristics PuC–UC reactors

		450			900			I350			2250		
Power density, kw/litre		300	500	700	300	500	700	200	400	600	200	400	600
Volume fractions	fuel	0.58	0.48	0.40	0.52	0.4I	0.30	0.55	0.40	0.26	0.5I	0.35	0.20
	sodium	0.25	0.32	0.37	0.30	0.38	0.46	0.29	0.40	0.49	0.33	0.46	0.57
	steel	0.I7	0.20	0.23	0.I8	0.2I	0.24	0.I6	0.20	0.25	0.I6	0.I9	0.23
Critical mass of plutonium,kg		860	535	4I0	I500	8I0	590	2860	I420	830	4600	I860	I030
BR		I.85	I.80	I.72	I.67	I.60	I.58	I.72	I.58	I.38	I.62	I.43	I.28
BRa		I.27	I.0	0.65	I.27	0.87	0.70	I.36	I.0I	0.68	I.37	0.98	0.63
U-238 in reactor core fission fraction		0.206	0.I62	0.I3I	0.I97	0.I43	0.II2	0.I98	0.I55	0.088	0.I82	0.I34	0.069
		0.I97	0.I88	0.204	0.2I5	0.2I7	0.224	0.2I5	0.230	0.242	0.227	0.243	0.260
The reactor core average		I.89	I.9I	I.93	I.9I	I.94	I.95	I.92	I.95	2.0I	I.94	I.98	2.04
		0.048	0.047	0.047	0.046	0.049	0.047	0.044	0.044	0.044	0.042	0.042	0.04I
		0.270	0.278	0.29I	0.28I	0.286	0.29I	0.287	0.296	0.3I4	0.296	0.3I2	0.326
		0.346	0.347	0.348	0.340	0.34I	0.346	0.345	0.347	0.35I	0.348	0.358	0.362
The blanket average		0.0I8	0.0I8	0.0I8	0.0I7	0.0I8	0.0I8	0.0I7	0.0I7	0.0I7	0.0I7	0.0I7	0.0I7
Doubling time, year T_{pr} = 0.5 y β = 8%		6.9	5.9	5.2	6.6	5.4	6.0	7.5	5.5	7.8	8.0	6.0	9.4

Table III

Parameters of power reactors with difference heat generation flattening
(core power 1000 Mw, PuC-UC fuel).

Flattening method	two "enrichment"			cylinder IB*		ring IB (10cm thickness)	
Reactor number**	1	2	3	4	5	6	7
Dense volume of reactor core (together with IB), Va, liters	1570	1000	1570	1170	1570	1110	1570
IB volume, % Va	-	-	-	7.8	6.3	9.2	7.8
Radial non-uniformity factor for fuel assemblies	1.7	1.21	1.24	1.24	1.28	1.18	1.26
Sodium fraction	0.39	0.39	0.32	0.39	0.33	0.33	0.33
Fuel fraction	0.38	0.38	0.47	0.38	0.46	0.38	0.46
Pu concentration, $\frac{N_9}{N_9 + N_8}$	0.127	0.13 and 0.17	0.11 and 0.15	0.161	0.125	0.168	0.128
Pu critical mass, kg	780	610	880	645	850	645	870
Specific power, Mw/kg Pu	1.28	1.64	1.12	1.55	1.18	1.55	1.15
Breeding ratio, BR	1.6	1.66	1.68	1.68	1.74	1.65	1.75
BRa	0.8	0.6	0.8	0.61	0.83	0.57	0.80
Doubling time T_n=0.5 years β=5%	4.8	3.8	4.7	3.8	4.6	4.0	4.4
T_n=1, β=5%	8.3	8.1	7.4	7.6	7.5	8.4	7.2

* IB - internal blanket

** reactor flattening plays to reduce the volume (NN 2,4,6) or fraction of sodium (NN 3,5,7).

Table IV

Power and temperature coefficients in oxide reactor
Va = 1800 liter, fuel - U-235

	Power coefficient 1/Mw x 10^5	Temperature coefficient 1/°C x 10^5
a) Doppler-effect	- 0.5	- 0.8
b) Fuel Element elongation	- 0.25	- 0.2
c) Assembly Divergence	- 0.03	- 0.3
d) Sodium Expansion	- 0.03	- 0.3
e) Fuel Element Bending	+ 0.02	-
Total	~ - 0.8	-2.2

Table V

Comparison of Various Fuel Cycles for an Oxide Reactor.
Reactor core output 1200 Mw, Power Density 400 kw/litre.
Volume fractions: Fuel-0.39, Sodium-0.43, Steel-0.18.

Cycle		Pu^{239}-U^{238}-Pu^{239}	U^{233}-Th^{232} U^{235}	Mixed
Critical mass, kg	Pu-239	1140	-	645
	U-233	-	1100	365
	Pu^{239}+U^{233}	1140	1100	1010
BR		1.50	1.17	1.35
BRa		0.69	0.56	0.65
U^{238} or Th fission fraction in the Reactor Core, %		12(U-238)	2(Th)	11(U-238)
Average Reactor Core Characteristics Parametres	σ_{f_9}	1.01	-	1.96
	σ_{f_3}	-	2.88	3.12
	ν_{μ_9}	2.36	-	2.39
	ν_{μ_3}	-	2.36	1.21
Doubling time, Years		6.9	19.7	10.8

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LIST OF REFERENCES

1. Abagyan L.P., Bazazyants N.O., Bondarenko I.I., Nikolayev M.N. Group Constants in Nuclear Reactor calculations. Gosatomizdat (1964).

2. Leipunskii A.I. et al. Research in Fast Reactor Physics. Pap. 2038. Geneva (1958).

3. Leipunskii A.I., Kazatshkovsky O.D., Pynchasik M.S. Atomic Energy. Vol.11, issue 4 (1961) 370.

4. Leipunskii A.I., Kazatshkovsky O.D., Pinchasik M.S., Krasnoyarov N.V., Poydo M.S. Fast Reactors on Sodium Coolant. A report submitted to the III-d Geneva conference (1964).

5. Bondarenko I.I. et. al. The influence of resonance cross-section structure on the spread and slow-down of neutrons in the mediums and resonance effects in fissionable elements. A report submitted to the III-d Geneva conference (1964).

6. Shikov S.B. Atomic Energy, vol.6, issue 2 (1959) 162.

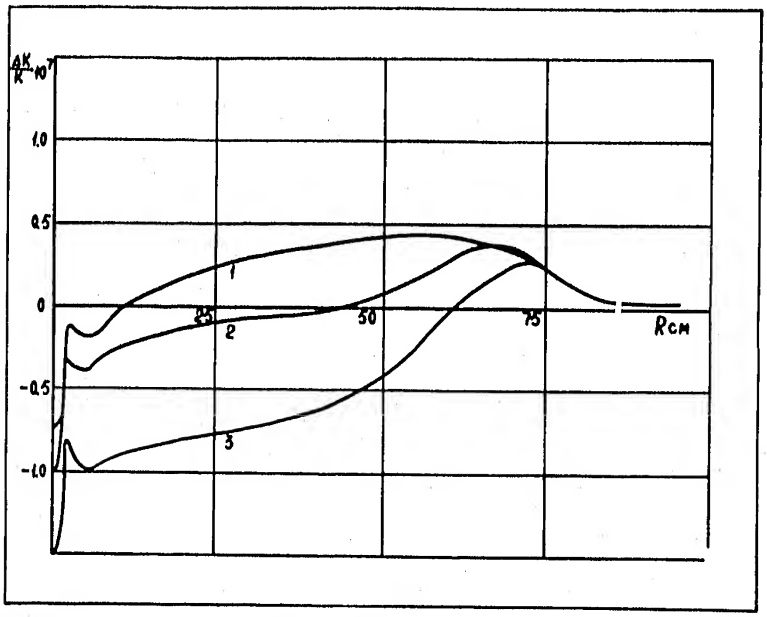

Fig. 1 Changes of the Oxide Reactor Reactivity of 1800 litre Volume with the Introduction of 1 cm^3 of Sodium (= 0.88g/cm^3). In the Centre of Reactor there is an absorbing Rod of the Efficiency of 0.3%. The reactor core radius is 75 cm.

1) Fuel- U^{235}, single enrichment. Sodium volume coefficient.

2) Fuel- U^{235}, two enrichments. $K_{Na} = -0.68 \cdot 10^{-5}\ 1/^{\circ}C$

$K_{Na} = -0.3 \cdot 10^{-5}\ 1/^{\circ}C$

3) Fuel - Pu, two enrichments, $K_{Na} = +0.16 \cdot 10^{-5}\ 1/^{\circ}C$.

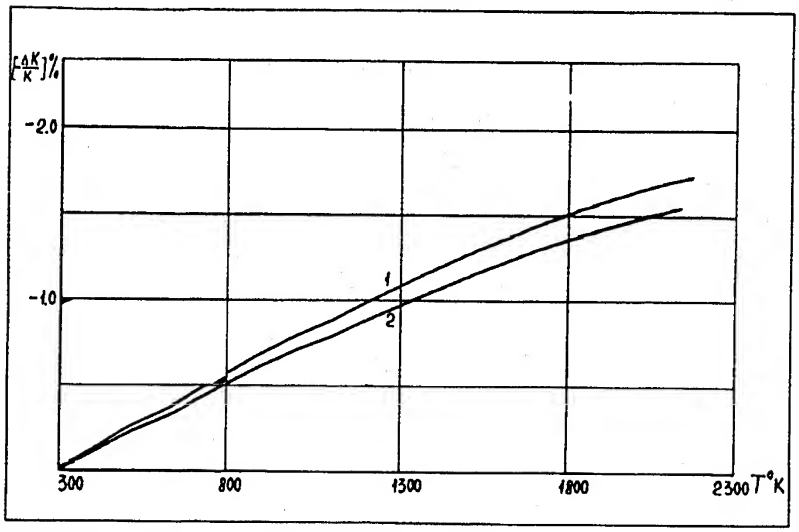

Fig. 2 Doppler-effect vs Temperature (Reactor with the Volume ~2000 litres; $PuO_2 - UO_2$)

1) Calculation for Perturbation Theory.

2) Direct Calculation

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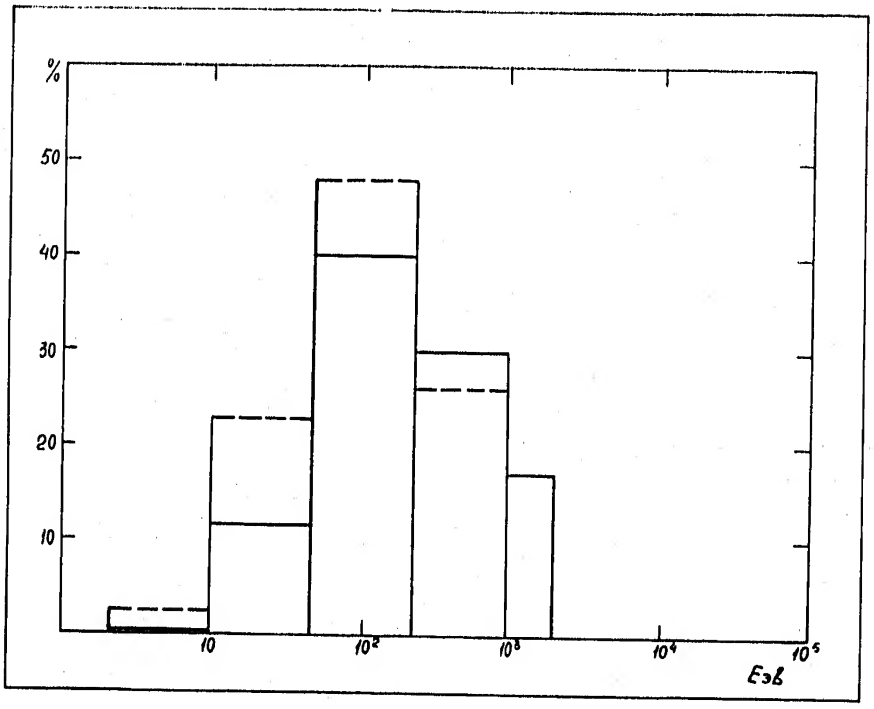

Fig. 3 Distribution of Doppler-effect along Energy Groups (% of the Total Effect of the Corresponding Isotope)

1) Fuel $PuO_2 - UO_2$, Va = 2000 l

2) Fuel - U^{238} (total effect - $1.3.10^{-5}$ $1/^oC$)

3) Fuel - Pu^{239} (total effect + $0.2.10^{-5}$ $1/^oC$).

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